NON-PUBLIC?: N
ACCESSION #: 9012030131
LICENSEE EVENT REPORT (LER)

FACILITY NAME: NORTH ANNA POWER STATION UNIT 2 PAGE: 1 OF 4

DOCKET NUMBER: 05000339

TITLE: REACTOR TRIP FROM 9% POWER DUE TO LOSS OF NORMAL FEEDWATER
EVENT DATE: 11/02/90 LER #: 90-010-00 REPORT DATE: 11/26/90

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 009

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(ii)

LICENSEE CONTACT FOR THIS LER:
NAME: G. E. Kane, Station Manager TELEPHONE: (703) 894-2101

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At 1741 hours, on November 2, 1990, Unit 2 experienced an automatic reactor trip from approximately 9 percent power. The reactor trip was caused by a low low level in "A" Steam Generator. The reactor trip occurred approximately eight minutes following an automatic trip of the turbine from approximately 15 percent power. This event is reportable pursuant to 10CFR50.73 (b) (2) (iv). A four hour report was made in accordance with 10CFR50.72 (b) (2) (ii).

The cause of the event was personnel error since failure to reset the feedwater regulating bypass valves after reinitiation of Main Feedwater resulted in the low low the Steam Generator level and the subsequent reactor trip. A contributing cause of the event was that a specific procedure for responding to a turbine trip without a reactor trip was not available. As an immediate corrective action, Emergency Procedure 2-E-0, Reactor Trip or Safety Injection, was entered and the

plant was stabilized. In addition, a root cause evaluation is being performed. Further corrective actions will be implemented, as required, based on the results of the root cause evaluation.

This event posed no significant safety implications because safety related equipment functioned as designed and key parameters stabilized following the reactor trip. There was no release of radioactive materials due to the reactor trip. The health and safety of the public were not affected at any time during this event.

END OF ABSTRACT

1.0 Description of the Event

At 1741 hours, on November 2, 1990, Unit 2 experienced an automatic reactor trip from approximately 9 percent power. The reactor trip was caused by a low low level in "A" Steam Generator (EIIS System Identifier AB, Component Identifier SG). The reactor trip occurred approximately eight minutes following an automatic trip of the turbine (EIIS System Identifier T, Component identifier TRB) from approximately 15 percent power. This event is reportable pursuant to 10CFR50 .73 (b) (2) (iv). A four hour report was made in accordance with 10CFR50 .72 (b) (2) (ii).

The Nuclear Regulatory Commission issued License Amendment Nos. 119 and 103, on July 18, 1989, for North Anna Units 1 and 2 , respectively. The amendments allowed the setpoint for direct reactor trip from a turbine trip to be increased from 10 percent power to 30 percent power. Full implementation of the amendments was to be completed following the completion of required modifications during the Unit 2 1990 and Unit 1 1991 refueling outages. North Anna Unit 2 fully implemented the amendment on October 30, 1990.

North Anna Unit 2 was returning to power operations from a scheduled refueling outage and Operations was preparing to place the unit on the electrical grid. When the Main Generator (EIIS System Identifier EL, Component Identifier GEN) was synchronized to the electrical grid, control room annunciators alarmed indicating there was a high level in "B" Steam Generator. Control room operators could not stabalize "B" Steam Generator level and a turbine trip occurred when the level reached the trip setpoint of 75 percent level. The Steam Generator high high level, by design, also caused the Main Feedwater Pump (EIIS Component Identifier P) to trip, Main

Feedwater isolation and the initiation of Auxiliary Feedwater (EIIS System Identifier BA). When Control Room Operators restarted the "B" Main Feedwater Pump, the Turbine Driven Auxiliary Feedwater Pump (EIIS Component Identifier P) automatically secured as designed, however, the Control Room Operators did not reset the feedwater regulating bypass valves (EIIS Component Identifier V) which resulted in a loss of feedwater flow to the "A" Steam Generator This resulted in a depletion of inventory in "A" Steam Generator. When "A" Steam Generator reached the low low level trip setpoint of 18 percent, a reactor trip occurred.

Control Room Operators responded to the reactor trip in accordance with procedure 2-E-0, Reactor Trip or Safety Injection. Reactor Coolant System (RCS) pressure decreased to approximately 2160 psig and RCS temperature decreased to approximately 534 degrees F. Pressurizer pressure and level were temporarily placed in manual control following the reactor trip to accelerate the return of these parameters to programmed values. Control was returned to automatic following stabilization.

No safety injection signal (manual or automatic) was initiated or required during the event. safety related and important equipment functioned as designed.

2.0 Significant Safety Consequences and Implications

This event posed no significant safety implications because safety related and important equipment functioned as designed and key parameters stabilized following the reactor trip. There was no release of radioactive materials due to the reactor trip. The health and safety of the public were not affected at any time during this event.

3.0 Cause of the Event

The cause of the event was personnel error since failure to reset the feedwater regulating bypass valves after reinitiation of main Feedwater resulted in the low low the Steam Generator level and the subsequent reactor trip. A contributing cause of the event was that a specific procedure for responding to a turbine trip without a reactor trip was not available.

4.0 Immediate Corrective Actions

As an immediate corrective action, Emergency Procedure 2-E-0, Reactor Trip or Safety Injection, was entered and the plant was stabilized.

5.0 Corrective Actions

Procedures for responding to a turbine trip without a reactor trip have been developed.

6.0 Additional Corrective Actions

Operators will be trained on the procedures for responding to a turbine trip without a reactor trip. A root cause evaluation will be performed. Further corrective actions will be implemented, as required, based on the results of the root cause evaluation.

7.0 Similar Events

A similar reactor trip occurred on December 5, 1989 and was reported in Unit 1 LER 89-017-00 dated December 28, 1989.

8.0 Additional Information

North Anna Unit 1 was in Mode 1 throughout this event and was not affected.

ATTACHMENT 1 TO 9012030131 PAGE 1 OF 1

Vepco VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION
P. O. BOX 402
MINERAL, VIRGINIA 23117 10 CFR 5
.73

November 26, 1990

U. S. Nuclear Regulatory Commission Serial No. N-90-021
Attention: Document Control Desk NAPS:JHL
Washington, D.C. 20555 Docket Nos. 50-339
License Nos. NPF-7

Dear Sirs:

The Virginia Electric and Power Company hereby submits the following

Licensee Event Report applicable to North Anna Unit 2.

Report No. 90-010-00

This Report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Corporate Management Safety Review Committee for its review.

Very Truly Yours

G. E. Kane
Station Manager

Enclosure:

cc: U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W.
Suite 2900 Atlanta, Georgia 30323

Mr. M. S. Lesser
NRC Senior Resident Inspector
North Anna Power Station

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